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Original Article

FEASIBILITY STUDY OF A DEDICATED NUCLEAR DESALINATION SYSTEM: LOW-PRESSURE INHERENT HEAT SINK NUCLEAR DESALINATION PLANT (LIND)

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ABSTRACT

In this paper, we suggest the conceptual design of a water-cooled reactor system for a low-pressure inherent heat sink nuclear desalination plant (LIND) that applies the safety-related design concepts of high temperature gas-cooled reactors to a water-cooled reactor for inherent and passive safety features. Through a scoping analysis, we found that the current LIND design satisfied several essential thermal–hydraulic and neutronic design requirements. In a thermal–hydraulic analysis using an analytical method based on the Wooton–Epstein correlation, we checked the possibility of safely removing decay heat through the steel containment even if all the active safety systems failed. In a neutronic analysis using the Monte Carlo N-particle transport code, we estimated a cycle length of approximately 6 years under 200 MW_{th} and 4.5% enrichment. The very long cycle length and simple safety features minimize the burdens from the operation, maintenance, and spent-fuel management, with a positive impact on the economic feasibility. Finally, because a nuclear reactor should not be directly coupled to a desalination system to prevent the leakage of radioactive material into the desalinated water, three types of intermediate systems were studied: a steam producing system, a hot water system, and an organic Rankine cycle system.

1. Introduction

A lack of fresh water is one of the most serious problems facing humans. The Middle East and Northern Africa region is well known as an area with water shortages. Because desalination technologies are attractive and sustainable solutions for this water crisis, desalination plants are being utilized to supply fresh water to people and industry. Multi-stage flashing (MSF), multi-effect distillation (MED), and reverse osmosis are commonly used for desalination processes.

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The target site of the nuclear desalination system proposed in this paper is the Abu Dhabi Emirate in the United Arab Emirates (UAE), which is located in the Middle East and Northern Africa region. The UAE, with almost 9 million people and US$44,000 of gross domestic product per capita in 2013, is one of the highest desalinating countries in the world (3682 m^3/d in 2013 [1]). The water resource demand is growing due to the expansion of agriculture and industry and the rapid population growth. This water demand will reach 5910 m^3/d by 2030 [1]. Currently, the UAE's desalination mostly depends on cogeneration fossil fuel plants, which utilize MSF, MED, or an electricity-based desalination process (reverse osmosis). Especially in Abu Dhabi, MSF is the most commonly used desalination process. Wibisono et al. performed an economic analysis for a dedicated nuclear desalination system and several fossil fuel desalination systems under the existing environmental and economic conditions in the UAE. The nuclear desalination option is cost competitive, compared with several fossil fuel options [2]. Because of the instability of fossil fuel prices and possibility of running out of fossil fuel resources, we expect that the dedicated nuclear desalination option will become more economical than fossil fuel desalination options in the future. Therefore, even though the UAE belongs to the Organization of Petroleum Exporting Countries, exporting fossil fuel and utilizing nuclear energy for desalination will be more economical than utilizing fossil fuel for desalination. Furthermore, since the exhaustion of fossil fuel is inevitable in the future, utilizing nuclear energy for desalination is a smart choice for the UAE from a long-term viewpoint. Jung et al. performed a feasibility study of a small-sized nuclear heat only plant dedicated to desalination in the UAE, and showed its potential in comparison with a desalination system based on a large nuclear power plant from the safety and economic perspectives [3].

Currently, water-cooled reactors are used in most of the world's nuclear power plants because they are the most proven technology and the most experience has been gained utilizing them. However, the Fukushima Daiichi nuclear accident, which occurred on March 11, 2011, showed the limitations of conventional light water reactors (LWR). Conventional LWRs have a high dependence on active systems and use Zircaloy cladding, which has unfavorable behavior at high temperatures. After the Fukushima accident, research on LWRs has focused on developing passive safety systems, along with melting- and chemical reaction-resistant silicon carbide (SiC) cladding.

Since the Fukushima accident, people have become more interested in high temperature gas-cooled reactors (HTGRs) because of their inherent and passive safety features. An HTGR can safely shutdown without control rods (CRs), and the decay heat can be removed without any active safety systems after shutdown under most accident scenarios. The following characteristics are the main reasons for the safe cool down: (1) the use of ceramic-coated particle fuel; (2) large decay heat removal capability of the reactor design due to the low power density core; and (3) annular graphite core with a high heat capacity and large surface area for heat transfer, which limits the peak fuel temperature during accidents. A safe reactor shutdown without CRs is possible because of the large negative temperature coefficient of reactivity and large temperature margins.

Although an HTGR has inherent and passive safety features, it is not as commercially proven as the LWR technology. Therefore, we applied the above-mentioned design characteristics of an HTGR to a water-cooled reactor for desalination. This allowed us to set the basic design concepts of the low-pressure inherent heat sink nuclear desalination plant (LIND) system. Much research has been conducted on coupling various nuclear reactors and desalination plants, and the low-pressure low-temperature reactor concept has already been proposed for a 200 MW_{th} nuclear heating reactor (NHR-200) and a reactor facility for heat supply with atmospheric pressure in the primary circuit (RUTA-70) [4,5]. However, the LIND reactor is the first nuclear reactor that applies the inherent and passive safety features of HTGRs to a water-cooled reactor.

Additionally, for a more realistic system design, we designed not only a primary system but also an intermediate system between the nuclear reactor and desalination plant. Three types of intermediate systems for the dedicated nuclear desalination system were studied: a steam producing system, a hot water system, and an organic Rankine cycle (ORC) system.

2. LIND system

In order to develop an inherently and passively safe water-cooled reactor for desalination, the basic design concepts and design requirements were set for the LIND system.

2.1. *Basic design concepts*

The LIND system adopts several innovative design concepts as shown in Fig. 1: (1) the use of low-pressure operation (1–3 bar) to lower the stored energy in the water and enhance the safety during a loss of coolant accident (LOCA), while maintaining sufficient pressure for dedicated desalination; (2) use of a pool-type reactor to have a large water reservoir; (3) use of square ring-type fuel loading (Fig. 2) and a low power density core to maintain the peak cladding temperature (PCT) below the criterion, even in an accident scenario; (4) use of SiC as the CR wall, baffle, guide tube, cladding, etc. to allow high-temperature components for decay heat removal; (5) use of SiC as cladding to permit critical heat flux during the worst transients and eliminate the possibility of H_2 explosion; (6) underground construction of a reactor to eliminate the concrete wall surrounding the steel containment for protection against any external clashes and for radiation shielding purposes; (7) use of a reactor cavity for ex-vessel cooling; (8) use of steel containment to effectively remove the decay heat through radiation and air convection, where the water condensed on the inner wall of the steel containment accumulates in the reactor cavity and cools the reactor vessel wall; and (9) use of an in-containment refueling water storage tank and desalted water storage tank as water storage for gravity-driven safety injection purposes.

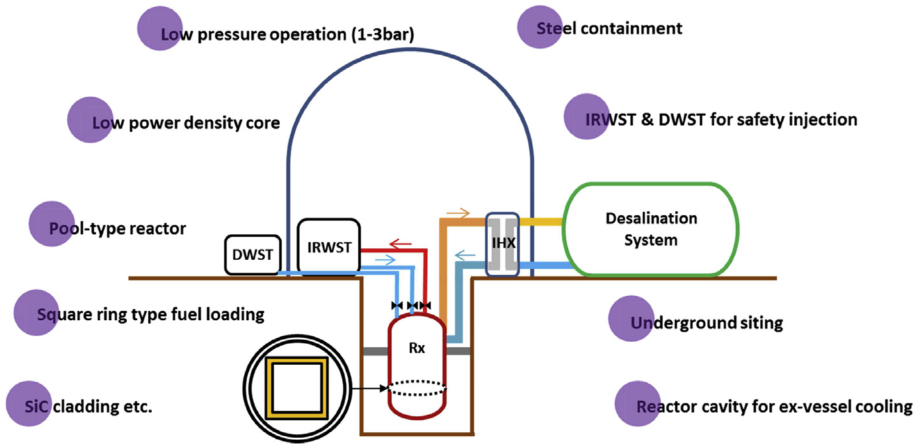

Fig. 1 – Basic design concepts of LIND system.

2.2. *Key design concepts*

The key characteristic of LIND is the sufficient decay heat removal capability even if all the active safety features fail during an extended station blackout (SBO) accident. In this study, we focused on an extended SBO with a LOCA. In the case of a non-LOCA event such as the failure of the intermediate heat transport system between the nuclear reactor and the desalination plant, if the decay heat cannot be removed by any safety system, it may expand into an LOCA event due to the pressure buildup on the primary side. The decay heat removal mechanism under the SBO + LOCA condition will be explained in Section 3.2.

2.3. *Design requirements*

The design requirements can be categorized into five types: (1) normal operating requirements; (2) thermal–hydraulic requirements; (3) neutronic requirements; (4) material requirements; and (5) coupling requirements. The normal operating requirements are as follows: (1) full reactor power, 100–300 MW_{th}; (2) reactor pressure, 1–3 bar; and (3) reactor outlet temperature, 90–120°C.

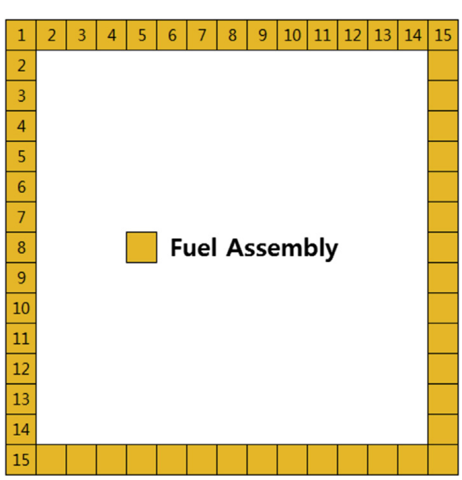

Fig. 2 – Core configuration of LIND system (top view).

Because the LIND system uses square ring-type fuel loading (Fig. 2) to maintain the PCT below the safety criterion during passive decay heat removal by radiation heat transfer, there is a limitation on the full reactor power. We attempted to increase the thermal power within a reasonable reactor vessel size. The LIND system adopts an MED desalination process, which requires a heat source at around 100°C. Thus, the primary side of the LIND system can be operated at a very low reactor pressure compared to conventional light water reactors for generating electricity. This low-pressure operation has advantages under accident conditions, including safety injection. However, it should be noted that the LIND system does not allow bulk boiling in the reactor vessel.

The thermal–hydraulic requirements are as follows: (1) sufficient decay heat removal capability through the containment by air cooling and radiation; (2) sufficient water capacity for the reactor vessel to remove decay heat for the first 3 days after shutdown; (3) an allowable design containment pressure of 5 bar; (4) an allowable reactor vessel wall temperature of <500°C; (5) allowable SiC cladding, SiC CR wall, and SiC baffle temperatures of < 1,300 ± α°C; (6) an allowable fuel temperature of < 2,865°C (incipient fuel melting temperature); and (7) during transients, a minimum departure from nuclear boiling ratio (MDNBR) of > 1.3. The first six requirements are related to the key design concepts mentioned in Section 2.2. In a serious accident scenario, in order to remove the decay heat safely, the LIND system should satisfy the above thermal–hydraulic requirements.

For the above thermal–hydraulic requirements, no specific temperature criterion for SiC materials had previously been specified. Therefore, we investigated several issues to determine the temperature criterion for SiC materials. Since the highest temperature in the SiC materials occurs on the SiC cladding surface, the temperature criterion for SiC materials is based on the research on SiC claddings.

Chemical vapor deposition β-SiC is one of the future candidates for cladding material. It is a very dense polycrystalline that is chemically stable, and has high strength and good resistance to radiation. The SiC diffusion coefficients are low enough to prevent releases of fission products. SiC is widely used in HTGR design in the form of tristructural–isotropic (TRISO) fuel. Small kernels of UO_2 fuel are surrounded by several layers of pyrolytic carbon and a protective layer of β-SiC. This is called TRISO-coated particle fuel. The silicon carbide layer in an HTGR is the strongest barrier to prevent the release of fission products. It effectively retains gaseous and metallic fission products, except silver [6]. The maximum allowable fuel temperature limit when using TRISO-coated particles is 1600°C in the current HTGR design. SiC decomposition occurs above 1600°C. The diffusion of fission products begins to increase above the assumed temperature limit [6].

SiC corrosion by high temperature vapor is a major challenge for SiC cladding in an LWR application. SiC is oxidized when it is exposed to vapor at > ~1100°C temperature. It forms a protective silica scale. However, this SiO_2 layer on the SiC layer is volatile at the same condition. The following equations show the dominant SiC oxidation reaction and volatilization reaction:

$$SiC + 3H_2O(g) = SiO_2 + 3H_2(g) + CO(g) \tag{1}$$

$$SiO_2 + 2H_2O(g) = Si(OH)_4(g) \tag{2}$$

Although some studies [7] have been performed, our current understanding of SiC oxidation and volatilization under high-temperature vapor conditions is limited. Moreover, the previous experiments were performed using a very slow vapor flow rate. The SiC layer recession was evaluated under a limited condition. Opila and Hann suggested that the linear volatility of SiO_2 was $2.07 \times 10^{-3} mgSiO_2/(cm^2h)$ at 1,300°C [7]. The typical density of SiC is 3.14 g/cm^3. It takes about 3,800 days for the entire SiC layer to become volatile if the thickness of the SiC cladding is 0.6 mm. By contrast, Lee et al. suggested a value of $4 \times 10^{-2} mgSiC/(cm^2h)$ at 1,140°C [8]. The SiC volatilization is predicted to be 200 days using a calculation based on the aforementioned data. One of the major issues with conventional cladding materials is the production of hydrogen by oxidation. The equivalent cladding reacted (ECR) is the ratio of the reacted cladding thickness to initial cladding thickness. The equivalent cladding reacted of SiC is approximately 1/5,000 that of Zr-4 [8]. This means that hydrogen production, one of the major safety issues, is negligible for SiC cladding compared to Zr-4 cladding.

As previously discussed, there are two temperature criteria for SiC cladding: (1) 1,600°C for SiC decomposition; and (2) 1,100°C for SiC corrosion by high-temperature vapor. The SiC cladding should not exceed 1600°C as a result of the diffusion of fission products. However, because SiC corrosion by steam is a very slow process, we can temporarily allow SiC corrosion by steam. Although it is a very slow reaction, a thin SiC cladding cannot endure for an infinite time.

Under an accident condition after reactor shutdown, the decay power decreases with time. As the decay power decreases, the PCT also decreases. Thus, if we can calculate the PCT as a function of the decay power and the recession thickness of the SiC cladding as a function of the PCT, we can determine the allowable peak cladding temperature after reactor shutdown based on the recession criterion. Therefore, we expect that the allowable PCT may be 1,300 ± α°C, since the limit falls in the range of 1,100–1,600°C.

The neutronic requirements are as follows: (1) sufficient reactor core reactivity and lifetime; (2) under-moderated core; and (3) boron-free operation and standby for shutdown. The material requirements are as follows: (1) reactor vessel material strength; and (2) boron-free operation and standby for shutdown. If the core temperature increases, the moderator temperature also increases, which produces a moderator density decrease. Under this condition, if the reactivity increases, the core is over-moderated. As the coolant temperature increases, the over-moderated core receives positive reactivity feedback. An under-moderated core has the opposite feature. Therefore, the reactor core of the LIND system should be under-moderated. Because boron can cause serious material problems in the system, it is one of the major concerns in system maintenance. The LIND system will use boron only during an accident scenario for an emergency reactor shutdown. Thus, the possibility of controlling the reactivity without boron was examined. It should be noted that the material requirements are not included in this paper but will be covered in other works. Finally, the coupling requirement is indirect coupling between the nuclear reactor system and desalination system to avoid radioactive material leakage to the desalinated water. Thus, we need to design an intermediate heat transport system for the LIND system.

3. LIND system analysis and results

3.1. *Main design parameters*

Table 1 lists the basic design parameters for the LIND system. These parameters were determined based on the thermal–hydraulic and neutronic system analyses discussed in the following sections. We attempted to satisfy the design requirements defined in Section 2.3 by optimizing the design concepts and design parameters. As a result, the following design parameters were selected.

3.2. *Thermal–hydraulic analysis of SBO + LOCA scenario*

As mentioned in Section 2.2, the key concept of the LIND system is to have sufficient decay heat removal capability

Table 1 – Basic design parameters of LIND system.

Core/Vessel		Fuel		Containment	
Q (MW_{th})	200	Assembly type	14 × 14	D (m)	15.7
P (MPa)	0.3	No. of Assembly	56	H (m)	27.9
T_{hot} (°C)	120	OD_{clad} (mm)	10.4394	$Q_{solar, max}$ (W/m^2)	1041
T_{cold} (°C)	75	ID_{clad} (mm)	8.89	$Q_{solar, avg}$ (W/m^2)	369
$D_{core, eff}$ (m)	3.11	Pitch (mm)	14.757		
D_{vessel} (m)	5.50	$L_{fuel\ active}$ (m)	3.81		
H_{vessel} (m)	12.81	q′ (kW/m)	5.3		

even if all the active safety features fail during an extended SBO accident. We designed the LIND system to allow it to be safely and naturally cooled down even in the case of a serious accident. We assumed an SBO + LOCA accident condition. First, the thermal–hydraulic requirements were checked using a simple analytical method. The ANS 2005 standard model (ANS FP + ^{239}U + ^{239}Np decay power curve) was selected as the decay power curve in this scoping analysis. The following equations were used for the analysis:

$$\frac{P(t_s)}{P_0} = -6.14575 \times 10^{-3} \ln t_s + 0.060157 \text{ for } 1.5 \le t_s \le 400\,\text{s} \tag{3a}$$

$$\frac{P(t_s)}{P_0} = 1.40680 \times 10^{-1} \times t_s^{-0.286} \text{ for } 400 < t_s \le 4 \times 10^5\text{s} \tag{3b}$$

where t_s is expressed in seconds. This is a reasonable approximation of the ANS FP + ^{239}U + ^{239}Np decay power curve with a relative error of about ± 6% [9].

3.2.1. *Analytical method*

There are two phases in the SBO + LOCA scenario, as shown in Fig. 3. In the first phase, the decay heat can be removed by the water in the reactor vessel right after shutdown. Because of the 3-day water capacity requirement for decreasing the decay heat level, which can be safely removed by radiation from the reactor core, the LIND system can effectively remove the decay heat during the first 3 days after shutdown. However, in this phase, a dramatic quantity of steam is generated. Thus, it is necessary to check whether the containment pressure exceeds the design limit of 5 bar.

In the second phase, the decay heat should be removed mainly by radiation heat transfer from the core to the reactor vessel wall because there is no water in the reactor vessel to remove it. The decay heat transferred from the core to the reactor vessel wall by radiation can be effectively removed by the water accumulated in the reactor cavity and then through the steel containment wall.

In the second phase, because the radiation heat transfer is very ineffective, we need to check the temperature criteria for the following components in the reactor vessel: (1) the reactor vessel wall; (2) baffle; (3) guide tubes; (4) cladding; and (5) fuel.

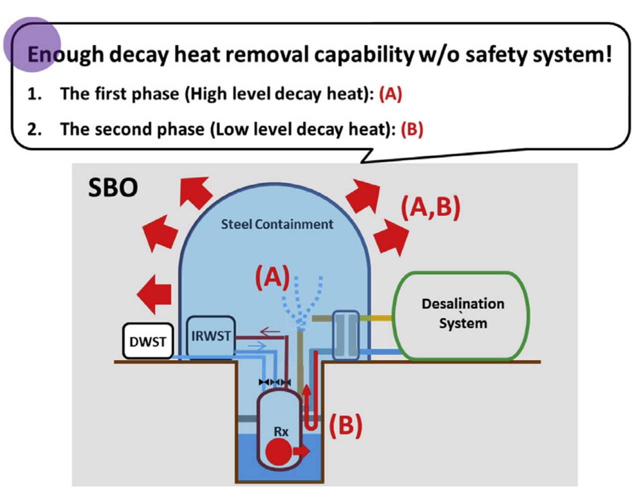

Fig. 3 – Decay heat removal mechanism in the station blackout + loss of coolant accident condition.

The water in the cavity makes it possible to maintain the vessel wall temperature at around 100°C. In the case of the fuel temperature, if the SiC cladding temperature does not exceed the PCT limit, the maximum fuel temperature will never exceed the incipient fuel melting temperature. This is because the decay heat level is very low (< 0.4% of full power) and the temperature difference between the center of the fuel and the cladding surface is very small, i.e., only several degrees Celsius in the second phase. In addition, in the SiC material components, the highest temperature occurs on the cladding surface. Therefore, in the second phase, our concern is whether or not the PCT exceeds the criterion. However, because the exact temperature criterion for the SiC cladding has not yet been decided, the estimated results for the PCT were not applied to the design parameter decision. In order to estimate the PCT, the surface-to-surface (S2S) radiation heat transfer equation suggested by Wooton and Epstein was used in the core part [10]:

$$Q_{mn} = \sigma A_m F_{mn} (T_m^4 - T_n^4)$$
$$T_m = \left(\frac{Q_{mn}}{\sigma A_m F_{mn}} + T_n^4 \right)^{1/4} \tag{4a}$$

$$F_{mn} = \left[\frac{1}{f_{mn}} + \frac{1}{\varepsilon_m} + \frac{A_m}{A_n} \left(\frac{1}{\varepsilon_n} - 1 \right) \right]^{-1} \approx \left[\frac{2}{\varepsilon_{m,n}} - 1 \right]^{-1} \left(\because f_{mn} = 1, \varepsilon_m = \varepsilon_n = 0.8, \text{ and } \frac{A_m}{A_n} \approx 1 \right) \tag{4b}$$

where there are 14 layers; the heat transferred by radiation from layer *m* to layer *n* is designated as Q_{mn}; A_m is the heat transfer area of layer m; the emissivity of layers *m* and *n*, $\epsilon_m = \epsilon_n$ is 0.8; the geometric view factor f_{mn} is 1.0 for concentric surfaces; and the ratios of successive radii, A_m/A_n, are close to one. Because the decay power decreases with time after reactor shutdown, the PCT was estimated at the beginning of the second phase.

In both phases, the steam generated in the vessel and cavity is mixed with air in the containment and condensed on the containment inner wall. Then, the condensed water accumulates in the cavity. Finally, the decay heat transferred to the steel containment wall is transferred to the environment by radiation and convection. The equation proposed by Abdullah and Karameldin was used to calculate the condensation heat transfer on the containment inner wall [11]. The Churchill–Chu correlation for an external flow across a vertical plate was used to calculate the convection heat transfer on the containment outer wall [12]. In the calculation of the radiation heat transfer from the containment outer wall to the environment, we considered the solar radiation at the target site, Abu Dhabi, UAE. In the calculations of the containment pressure buildup history and temperature distribution of the LIND system, we used the highest daily mean solar radiation (369 W/m^2) and the highest 1-minute average daily solar radiation (1041 W/m^2), respectively (Table 1) [13].

Figs. 4 and 5 show the analytical results. The maximum containment pressure was estimated to be 3.75 bar (Fig. 4) in the first phase. The reactor vessel wall temperature was almost the same as the cavity water temperature as a result of the very low level of decay heat flux. Therefore, there was

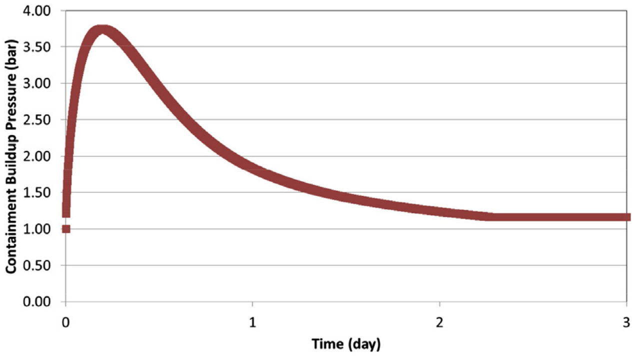

Fig. 4 – Containment pressure buildup history.

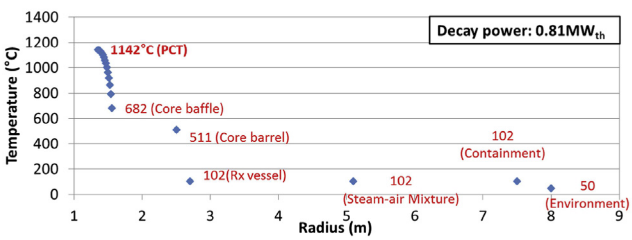

Fig. 5 – Temperature distribution in LIND system.

a large margin for the reactor vessel wall temperature. The PCT was estimated to be 1,142°C at the beginning of the second phase, 3 days after reactor shutdown (Fig. 5). The decay heat level at 3 days after shutdown was 0.809 MW$_{th}$ (0.4045% of the total power, 200 MW$_{th}$). In the scoping analysis, the maximum containment pressure and reactor vessel wall temperature satisfied each criterion. In the case of the PCT, although the temperature criterion for the SiC cladding has not yet been exactly defined, the analytical result (1,142°C) was very good compared to the current allowable PCT criterion (1,300±α°C).

3.3. *Reactor core thermal–hydraulic analysis in normal operation*

3.3.1. *Allowable surface heat flux*

In the normal operating condition, we checked whether the actual heat flux had a sufficient margin for the critical heat flux. The MDNBR was used as an index for this. Table 2 lists the normal operating conditions of the LIND system. The Bowring correlation was used to estimate the MDNBR value [14]:

Table 2 – Normal operating condition of LIND system.

Total thermal power	200 MW
Rod power	20.3 kW
Peak heat flux	255 kW/m^2
Coolant inlet temperature	75°C

$$q''_{cr} = \frac{A - Bh_{fg}x}{C} \tag{5a}$$

$$A = \frac{2.317(h_{fg}DG/4)F_1}{1 + 0.0143\,F_2D^{1/2}G} \tag{5b}$$

$$B = \frac{DG}{4} \tag{5c}$$

$$C = \frac{0.077F_3DG}{1 + 0.347F_4\left(\frac{G}{1356}\right)^n} \tag{5d}$$

$$p_R = 0.145p(\text{where } p \text{ is in MPa}) \tag{5e}$$

$$n = 2.0 - 0.5\,p_R \tag{5f}$$

For $p_R < 1$ MPa:

$$\left.\begin{aligned} F_1 &= \{p_R^{18.942}\exp[20.891(1-p_R)] + 0.917\}/1.917 \\ F_2 &= F_1/(\{p_R^{1.316}\exp[2.444(1-p_R)] + 0.309\}/1.309) \\ F_3 &= p_R^{17.023}\exp[16.658(1-p_R)] + 0.667\}/1.667 \\ F_4 &= F_3/p_R^{1.649} \end{aligned}\right\} \tag{5g}$$

For $p_R > 1$ MPa:

$$\left.\begin{aligned} F_1 &= p_R^{-0.368}\exp[0.648(1-p_R)] \\ F_2 &= F_1/\{p_R^{-0.448}\exp[0.245(1-p_R)]\} \\ F_3 &= p_R^{0.219} \\ F_4 &= F_3/p_R^{1.649} \end{aligned}\right\} \tag{5h}$$

The Bowring correlation is applicable under the following conditions:

$D = 0.002 - 0.045$ m
$L = 0.15 - 3.7$ m
$p = 0.2 - 19.0$ MPa
$G = 136 - 18,600$ kg/m^{2}s

Our system conditions were as follows:

$D_e = 0.01612$ m
$L = 3.81$ m
$p = 0.3$ MPa
$G = 524.5$ kg/m^{2}s

These were fairly applicable. Fig. 6 shows the DNBR values in comparison to the axial position. The MDNBR was estimated to be 36.9 coming from a very low power density compared with that of conventional LWRs.

3.3.2. *Core axial temperature distribution*

In the thermal–hydraulic analysis, we finally estimated the core axial temperature distribution in order to utilize the temperature information in the reactor neutronic analysis. Because of the uncertainty of the gap conductance between the fuel and cladding, and the thermal conductivity of the fuel and cladding, we considered two extreme cases. The lower-bound temperature condition assumed: (1) no gap between the fuel and cladding due to fuel expansion; and (2) no irradiated fuel and cladding [15–17]. We obtained the lowest axial temperature distribution under these conditions. The upper-bound temperature condition assumed: (1) a gap filled with helium; and (2) irradiated fuel and cladding [15–17]. The highest axial temperature distribution was obtained under

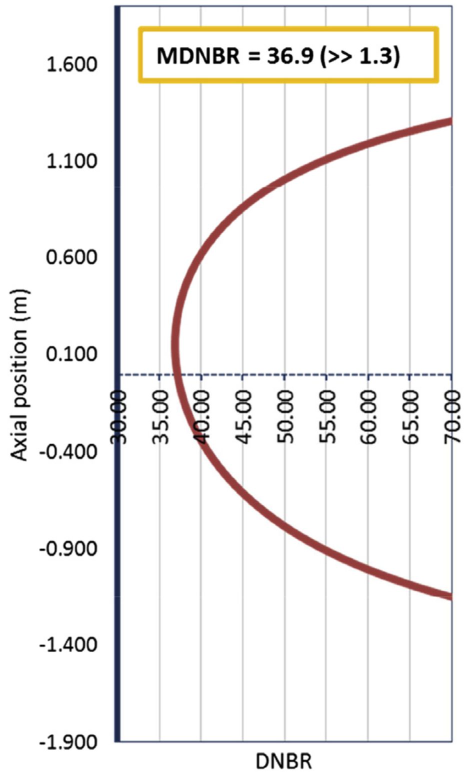

Fig. 6 – Departure from nuclear boiling ratio (DNBR) versus axial position in normal operation.

these conditions. Figs. 7 and 8 show the axial temperature of the fuel center and surface, temperature of the cladding's inner and outer surfaces, and bulk temperature of the coolant, as functions of the distance along a coolant channel.

3.4. *Reactor core neutronic analysis*

3.4.1. *Determination of fuel enrichment*

Although we have verified the feasibility of the LIND system from a thermal–hydraulic viewpoint, we also needed to confirm the feasibility of operating the LIND core. Therefore, we estimated reactor cycle lengths for several enrichments and the reactivity control capability in a fresh core without chemical shim control. First, depletion calculations for several enrichments were performed to check whether the cycle length was long enough.

The depletion calculations were performed using the MCNP5 [18], MONTEBURNS 1.0 [19], and ORIGEN 2.2 [20] code systems. Fig. 9 shows the MCNP5 input geometry, while Tables 3 and 4 list the calculation conditions for MCNP5 and MONTEBURNS 1.0, respectively. The LIND core was divided into five axial nodes to apply the axial temperature distribution under the lower bound condition, as described in Table 5. Cross section libraries were processed using NJOY99 [21] to consider the temperature effect on the cross sections.

The depletion calculation results are shown in Fig. 10 and listed in Table 6. Fig. 10 shows the cycle lengths for three enrichment candidates (4.5 weight percent (w/o), 9.0 w/o, and 19.9 w/o), while Table 6 lists the discharge burnups along the axial node for each enrichment case. For the 4.5 w/o enriched fuel, the cycle length is around 6 years, and the average discharge burnup is around 26.4 GWd/MTU.

To eliminate boric acid, which is used for chemical shim control and causes material degradation, CRs must control the excess reactivity in a fresh core. Thus, it was necessary to verify that the CRs could suppress the initial excess reactivity for the three enrichment candidates.

The specifications of the CRs are listed in Table 7. CRs composed of B_4C and SiC were inserted in every guide thimble except for a center thimble of each assembly. The effective multiplication factors (k_{eff}) were calculated for the CR-in and CR-out cases in the fresh core. Table 8 lists the MCNP5 calculation conditions. The calculation results listed in Table 9 indicate that none of the three enrichment candidates could achieve subcriticality with CRs. Thus, burnable absorbers should be inserted in the core to reduce the initial excess reactivity.

The specifications of these burnable absorber rods are shown in Fig. 11. For each assembly, 16 burnable absorber rods shown in green were inserted near the guide thimbles.

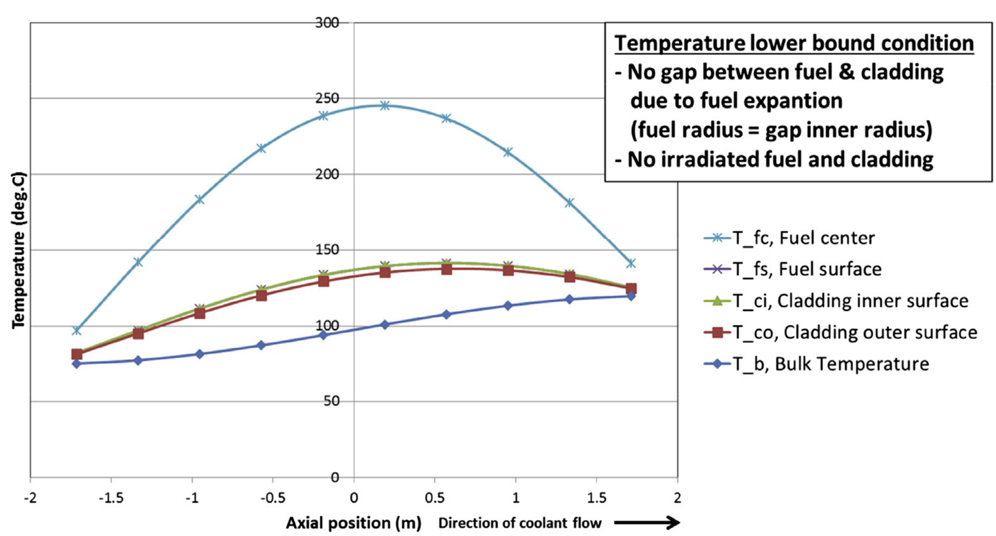

Fig. 7 – Core axial temperature distribution in lower bound condition.

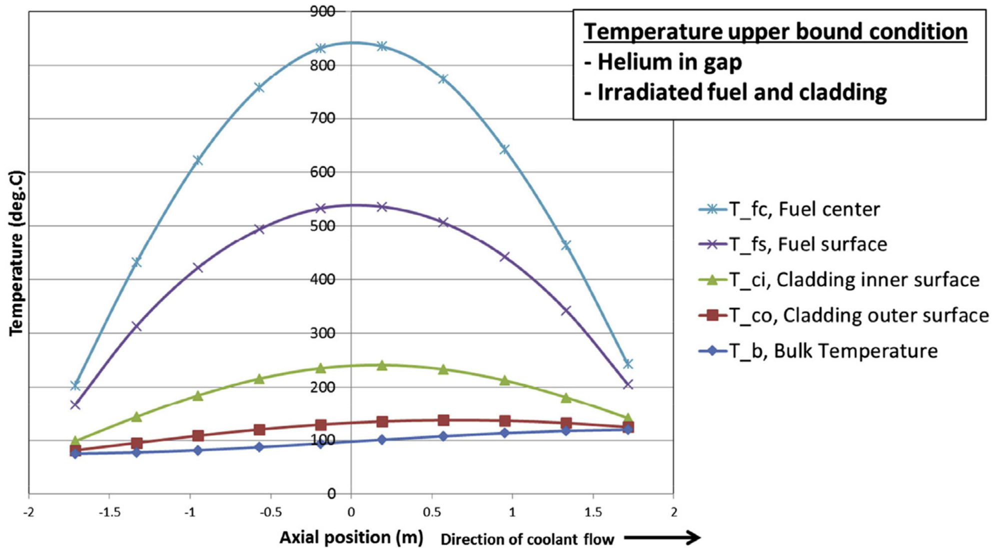

Fig. 8 – Core axial temperature distribution in upper bound condition.

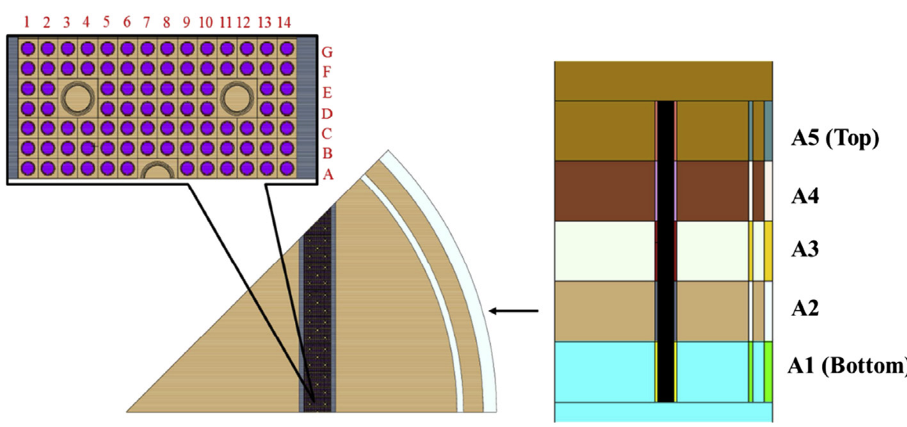

Fig. 9 – Horizontal cut (left) and vertical cut (right) of 1/8 LIND core.

Because the moderating power was large near the guide thimbles, the power distributions were flattened by locating the burnable absorber rods near the guide thimbles.

The k_{eff} values of the CR-in and CR-out cases were calculated for the LIND core with burnable absorber rods. The calculations were performed using MCNP5 with the calculation conditions listed in Table 8. As listed in Table 10, only the 4.5 w/o enrichment case could achieve subcriticality in the fresh core using CRs. From these results, the enrichment of the LIND core was determined to be 4.5 w/o.

Table 3 – MCNP5 calculation conditions for depletion calculations.

No. of history per cycle	20,000
No. of inactive cycle	50
No. of active cycle	200
Library	ENDF/B-VII.0

3.4.2. *Coolant void reactivity analysis*

When the coolant void reactivity is positive, we cannot guarantee reactor safety during an LOCA or other types of accidents. To confirm the negative coolant void reactivity of the LIND, we calculated the k_{eff} values of the 4.5% enriched fresh core (lower bound case) for several coolant densities. Note that the moderator density at 100% indicates the moderator

Table 4 – MONTEBURNS1.0 calculation conditions for depletion calculations.

Power	$25\ MW_{th}$	
Outer burn steps	1st–4th step	5 days
	5th–6th step	10 days
	7th step	20 days
	8th step	40 days
	9th step ~	50 days
Internal burn steps	400	

Table 5 – Temperatures of five axial nodes in lower bound condition.

Axial node	Coolant (°C)	Cladding (°C)	Fuel (°C)
A5 (top)	118.64	129.13	139.43
A4	110.56	138.86	166.59
A3	97.50	134.39	168.75
A2	84.44	116.13	143.09
A1 (bottom)	76.36	88.98	98.77

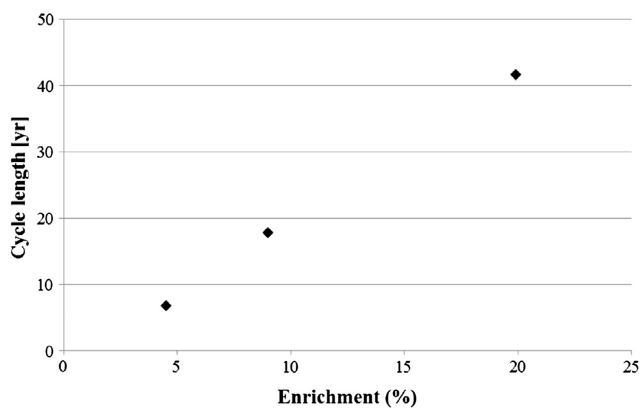

Fig. 10 – Cycle lengths of three enrichments (4.5 w/o, 9.0 w/o, 19.9 w/o).

Table 6 – Discharge burnups along five axial nodes for three enrichments.

Axial node	Discharge burnup (GWd/MTU)		
	4.5 w/o	9.0 w/o	19.9 w/o
A5 (top)	23.80	65.24	160.78
A4	28.02	70.59	168.73
A3	28.17	69.48	167.84
A2	28.26	70.41	168.30
A1 (bottom)	23.91	65.82	161.56
Average	26.43	68.31	165.44

Table 7 – Specification of control rods.

Absorber material	B_4C
Cladding material	SiC
Control rod pin radius (cm)	0.96266
Cladding inner radius (cm)	0.97121
Cladding outer radius (cm)	1.04868
Total No. of control rods	224 (4 rods per assembly)

density under the normal operating condition. The calculation conditions are listed in Table 11, and the calculation results are shown in Fig. 12. We can see that as the coolant density decreases, k_{eff} decreases, which indicates a negative coolant void reactivity. Thus, the LIND core is under-moderated.

3.5. *Intermediate heat transport system design*

The main purpose of the LIND reactor is supplying thermal energy for a seawater desalination process. A nuclear reactor should not be directly coupled to a desalination system to avoid radioactive material leakage to the desalinated water. A heat exchanger is needed to transfer the thermal power from the primary system to the intermediate system. This heat exchanger can be a steam generator or intermediate heat exchanger (IHX), depending on the operating conditions of the primary system and requirements of the desalination system. The primary system operating conditions used for designing the intermediate system for the LIND reactor are listed in Table 12.

Table 8 – MCNP5 calculation conditions for k_{eff} of CR-in and CR-out cases.

No. of history per cycle	30,000
No. of inactive cycle	200
No. of active cycle	500
Library	ENDF/B-VII.0

Table 9 – k_{eff} of CR-in and CR-out cases w/o burnable absorber rods.

Fuel enrichment (w/o)	CR-out case	CR-in case
4.5	1.21581	1.01832
9.0	1.31917	1.13123
19.9	1.39396	1.22471

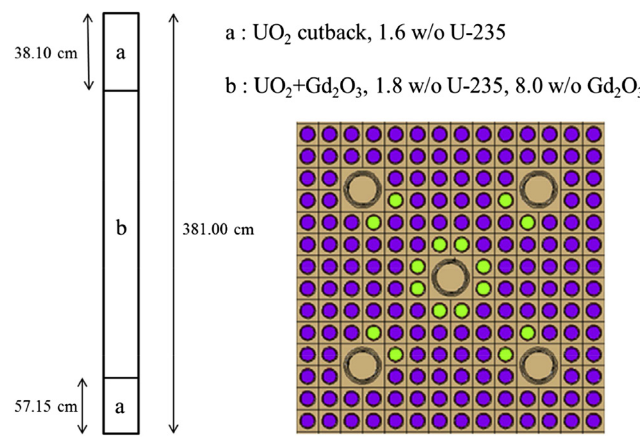

Fig. 11 – Axial material compositions (left) and positions (right, colored by green) of burnable absorber rods.

Table 10 – k_{eff} of CR-in and CR-out cases with burnable absorber rods.

Fuel enrichment (w/o)	CR-out case	CR-in case
4.5	1.13680	0.91693
9.0	1.24092	1.02222
19.9	1.32022	1.11409

Table 11 – MCNP5 calculation conditions.

No. of history per cycle	30000
No. of inactive cycle	200
No. of active cycle	500
Library	ENDF/B-VII.0

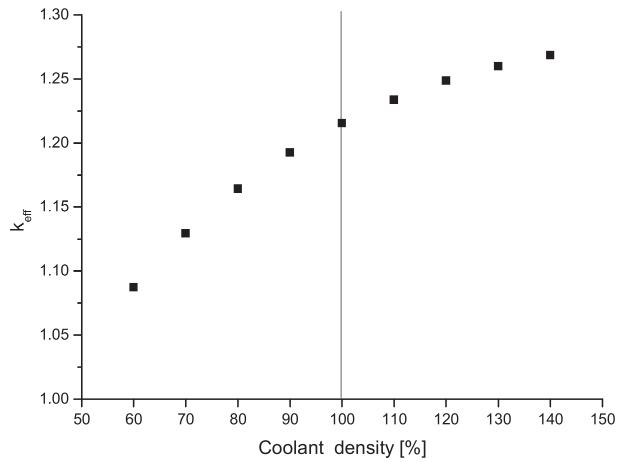

Fig. 12 – k_{eff} versus moderator density.

Table 12 – Primary system operating conditions for designing intermediate system.

Parameter	Value
Thermal power (MW_{th})	200
Operating pressure (MPa)	0.3
Reactor inlet temperature (°C)	75–90
Reactor outlet temperature (°C)	~120

The desalination system considered in this study uses the MED process. A requirement for the MED process is a top brine temperature for the first effect of approximately 70°C to prevent seawater scaling [22]. The general MED process usually utilizes low-pressure steam to increase the brine temperature to the required value (70°C) before entering the first effect of the MED process. For the LIND intermediate system design, three heat source options for the MED were considered: steam, hot water, and organic vapor.

3.5.1. *Steam option*

To utilize steam as a heat source for the MED process, a steam generator is needed. Since the reactor coolant outlet temperature is 120°C and the top brine temperature is 70°C, the steam produced in the intermediate system should be in the range of 70–120°C. To avoid scaling, it is preferable to utilize steam at a temperature closer to 70°C, which might require a low pressure condition for the steam (vacuum pressure). One way to achieve this is generating steam at a higher pressure and then lowering the operating condition to the target value. However, the low condition of the reactor coolant outlet temperature becomes a major problem when producing steam in this way. Another way to produce low-pressure steam is using a combination of an IHX and a flash tank, as shown in Fig. 13.

The problem with this coupling scheme is the log mean temperature difference (ΔT_{lm}) of the IHX is too small at 13.86°C. Even if we reduce the intermediate coolant outlet temperature, there will be no significant increase in the log mean temperature difference. The low ΔT_{lm} will require a high heat transfer surface area for the IHX. Another problem with this option is the inability to provide a pressure barrier for the leakage of radioactivity. Because the pressure of the intermediate system is lower than that of the primary system, radioactive material can leak into the intermediate system if a tube rupture accident occurs in the IHX.

3.5.2. *Hot water option*

The second option for the intermediate heat transport system design of the LIND system is utilizing the sensible heat of water to deliver thermal energy from the primary side to the MED system. The only equipment needed for this option is an IHX. Thus, it can be simpler than the steam production option. The flow diagram of this option is shown in Fig. 14. The intermediate system will utilize sensible heat from a single-phase water coolant by increasing its temperature from 60°C to 80°C at the IHX. The major drawback of this option is that the mass flow is quite high compared to the steam option. Therefore, a higher pumping power might be needed for this

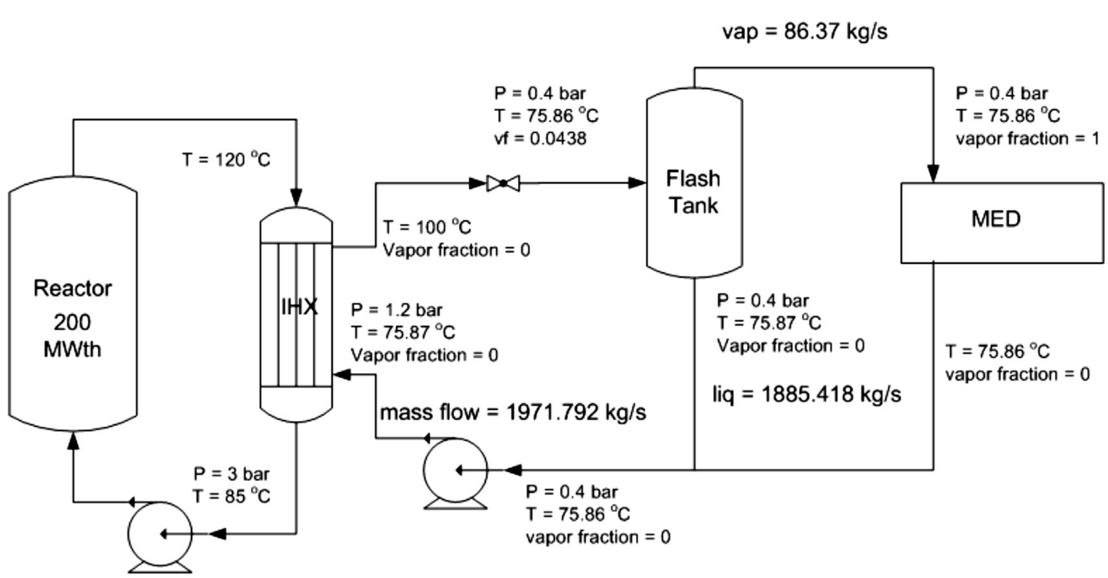

Fig. 13 – Intermediate heat transport system with low-pressure steam production.

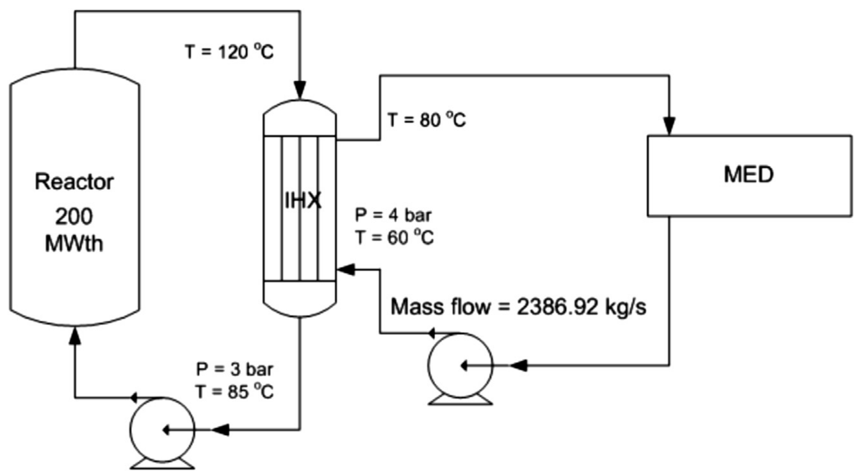

Fig. 14 – Intermediate heat transport system utilizing water sensible heat.

option compared to the steam generation option, but on the other hand they might be comparable because a flash tank and larger IHX are needed in the steam generation option. Another advantage provided by this option is the ability to operate the intermediate system at a higher pressure than the primary system, which provides a pressure barrier against radioactivity leakage if a tube rupture accident occurs in the IHX. In addition, it has been proven that hot water above 60°C can be used as the heat source for an MED plant.

A once-through type exchanger and U-tube type exchanger were considered for this option. The equipment was sized by following the procedure given in [23]. The sizing results for both types are listed in Table 13, and show that the U-tube type exchanger is better than the once-through type in terms of the size and pressure drop.

3.5.3. *Organic vapor option*

The idea of utilizing organic vapor was proposed to allow the system to produce a small amount of electricity in addition to the thermal power for desalination. Although MED is a thermal desalination process, it still requires some quantity of electric energy. The idea is to use ORC technology to produce a small amount of electricity to supply the plant's electricity needs and thus reduce the cost of purchased electricity. The ORC can be used because an organic vapor can have a boiling point lower than water with the same operating pressure. The MED energy requirement was calculated using the desalination economic evaluation program (DEEP) software developed by the International Atomic Energy Agency, and is listed in Table 14 [24].

The coolant selection for the ORC has to consider several points such as the operating condition, industrial experience, and environmental effects. Saleh et al. compared the use of various working fluids for the ORC [25]. Based on the LIND operating conditions, the most appropriate coolants for the ORC would be 1,1,2,2,3-pentafluoropropane (R245ca), 1,1,1,3,3-pentafluoropropene (R245fa), butane (R600), and n-hexane. The concept is to exchange the heat from the primary system using an organic coolant to produce an organic vapor at a superheated state and then use the ORC turbine to produce a small amount of electricity. The waste heat from the turbine is used to supply the thermal energy for the MED through a condensation process, after which the condensate is pumped back to the organic steam generator. The system was designed by using the design assumptions listed in Table 15 [26], and the process flow diagram of the

Table 13 – Comparison of once-through type exchanger and U-tube type exchanger for IHX.

Parameter	Once-through	U-tube
Log mean temp. diff., ΔT_{lm} (°C)	31.91	27.79
Overall heat transfer coeff., U (W/m^2-K)	941.52	1028.15
Heat transfer surface area (m^2)	6,655.94	6,999.83
Tube outer diameter (mm)	30	30
Tube wall thickness (mm)	3.2	3.2
Tube pitch (mm)	37.5	37.5
Tube length (m)	13	12
Number of tubes	5433	6110
Shell diameter (m)	2.9	3.08
Tube side velocity (m/s)	0.6	1.06
Shell side velocity (m/s)	1.46	1.29
Tube side pressure drop (kPa)	2.60	15.97
Shell side pressure drop (kPa)	185.90	138.87

Table 14 – MED energy requirements.

Parameter	Value
Water capacity (m^3/day)	20,000
Thermal energy needs (MW_{th})	56
Electricity needs (MW_e)	1.2

Table 15 – ORC system design assumptions.

Variable	Value
Turbine isentropic efficiency	75%
Turbine mechanical–electrical efficiency	95%
Pump isentropic efficiency	80%
Pump mechanical–electrical power efficiency	84.72%

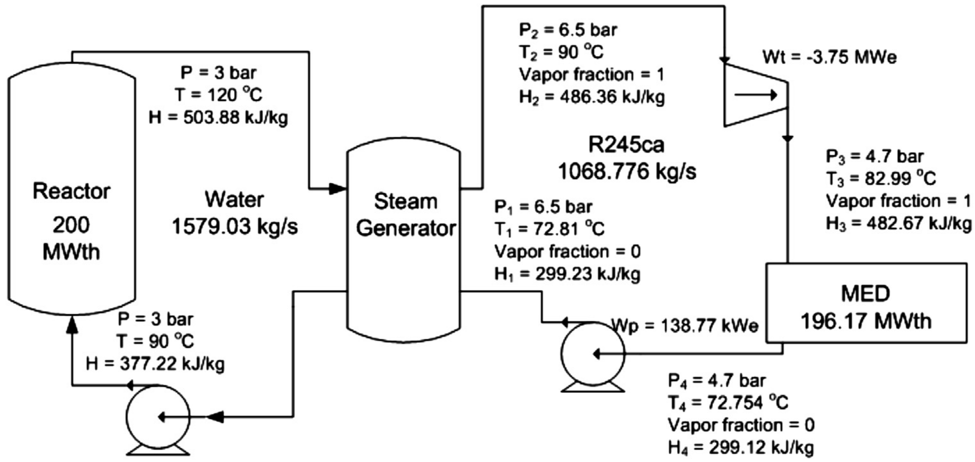

Fig. 15 – Intermediate heat transport system utilizing organic Rankine cycle.

organic vapor option using R245ca as a coolant is shown in Fig. 15. Similar to the hot water option, the organic vapor option also has the ability to provide a pressure barrier to reduce the chance of radioactive material leakage to the desalination plant. Among the four organic fluids proposed, only n-hexane will fail to provide the pressure barrier needed to prevent radioactive leakage from the primary system. The system design results for all the coolants are listed in Table 16.

4. Conclusions

A dedicated water-cooled nuclear desalination system (LIND) equipped with inherent and passive safety features was proposed and discussed. The target site of this plant is the Abu Dhabi Emirate in the UAE. In order to achieve the goal, we applied some of the safety-related design concepts of HTGRs to water-cooled reactors to provide inherent and passive safety features. The LIND adopts several innovative design concepts such as the use of square ring-type fuel loading and SiC cladding. The key characteristic of the LIND is the provision of a sufficient decay heat removal capability even if all the active safety features fail during an extended SBO accident. By performing a scoping analysis, we found that the current LIND design satisfied the essential thermal–hydraulic and neutronic design requirements, and the reactor core could be safely cooled down even during the SBO + LOCA scenario. In a thermal–hydraulic analysis using an analytical method based on the Wooton–Epstein correlation, we verified that the decay heat could be removed safely through the steel containment even if all the active safety systems failed. Based on the results of a reactor core neutronic analysis using the MCNP code, we estimated that the cycle length of the LIND core would be around 6 years under 200 MW_{th} and 4.5% enrichment. The very long cycle length and simple safety features minimize the burdens from operation, maintenance, and spent-fuel management, with a positive impact on economic feasibility. Finally, three types of intermediate systems for the dedicated nuclear desalination system were studied: a steam producing system, a hot water system, and an ORC system. The steam generating option provided the smallest log mean temperature difference (ΔT_{lm}) compared to the other options, but was not able to provide a pressure barrier against radioactive leakage. The organic vapor option had an advantage due to its ability to provide electricity. However, in term of the IHX size and coupling experience with an MED plant, hot water is the best option among the three proposed.

Table 16 – ORC system design parameters.

Parameter	R245ca	R600	R245fa	n-hexane
P_{max} (bar)	6.50	11.50	9.00	1.70
P_{min} (bar)	4.70	8.50	6.50	1.20
mass flow (kg/s)	1068.78	603.30	1083.07	541.81
T_1 (°C)	72.81	72.33	72.53	74.22
T_2 (°C)	90.00	90.00	100.00	95.00
T_3 (°C)	82.99	81.68	92.61	89.70
T_4 (°C)	72.75	72.16	72.41	74.20
Thermal supply (MW_{th})	196.17	196.02	196.11	195.67
Electricity produced (MW_e)	3.75	4.04	3.87	4.15
Pump work (MW_e)	0.14	0.33	0.22	0.05
ΔT_{lm} (once through)	23.00	23.29	18.71	20.04
Heat transfer surface area (m^2)	16960.04	14799.64	16074.32	32021.94

R245ca, 1,1,2,2,3-pentafluoropropane; R245fa, 1,1,1,3,3-pentafluoropropene; R600, butane.

5. Recommendations

In the LIND system, the key component is the SiC cladding. However, SiC cladding has not yet been commercially proven. In this paper, although we briefly mentioned SiC corrosion issues under very high temperature conditions, there are several other issues for SiC cladding, including the following: (1) a brittle fracture characteristic; (2) low thermal conductivity; (3) hermetic joining; and (4) fabrication. Based on general studies of the various issues, it is necessary to propose a specific temperature criterion for SiC cladding. In this paper, we focused on the technical aspects. In the next step, an economic evaluation needs to be performed.

Abbreviations

CR	control rod
HTGR	high temperature gas-cooled reactor
IHX	intermediate heat exchanger
k_{eff}	effective multiplication factor
LIND	low-pressure inherent heat sink nuclear desalination plant
LOCA	loss of coolant accident
LWR	light water reactor
MCNP	Monte Carlo N-particle transport code
MDNBR	minimum departure from nuclear boiling ratio
MED	multieffect distillation
MSF	multistage flashing
ORC	organic Rankine cycle
ORIGEN	Oak Ridge isotope generation and depletion
PCT	peak cladding temperature
SBO	station blackout
SiC	silicon carbide
TRISO	tristructural–isotropic
UAE	United Arab Emirates
w/o	weight percent

Conflicts of interest

All contributing authors declare no conflicts of interest.

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